CANDU PHYSICS CONSIDERATIONS FOR THE DISPOSITION OF WEAPONS-GRADE PLUTONIUM

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ABSTRACT

At the request of the US Department of Energy AECL has examined the feasibility of using CANDU® for the disposition of weapons grade plutonium. Utilizing existing CANDU technology, the feasibility of using MOX fuel in an existing CANDU reactor was studied. The results of this study indicate that the target disposition for disposal of weapons grade plutonium can be met without the requirement of any major modifications to existing plant design.

INTRODUCTION

In the course of investigating possible means for the disposition of weapons-grade plutonium, the US Department of Energy asked AECL to study the feasibility of burning plutonium in the form of MOX fuel in existing CANDU reactors. The spent fuel produced by the MOX cycle would be, upon exit from a CANDU reactor, similar to natural uranium spent fuel. Due to its radioactivity, this would make it as unattractive as current stockpiles of spent fuel from CANDU reactors. The 4 x 769MW(e) Bruce A generating station was chosen as a reference site. The target for disposal rate was set at 50 tonnes of plutonium over 25 years. This study will demonstrate that this can be accomplised using two of the four Bruce A reactors.

Two bundle designs were used in the study. One was the standard CANDU 37-element fuel bundle. The plutonium, combined with depleted uranium, is contained in the outer 30 fuel pins of the bundle. The inner 7 pins contain depleted uranium and a burnable poison to offset the excess reactivity of the plutonium fuel.

Additionally, an alternate fuel bundle design utilizing the CANFLEX bundle was studied. This bundle design, currently being qualified for commercial use, has 43 fuel elements. The elements in the two outer rings are of slightly smaller diameter than those in the two inner rings. This allows the elements to operate at lower ratings compared to the 37-element design, for the same bundle power. As a result of this lower power rating, the MOX elements, coupled with other internal element design optimizations, can achieve higher burnup without increasing the risk of fuel failures. In turn, this permits higher concentrations of plutonium to be used, significantly reducing the fuelling rate and thus reducing the fuel fabrication costs.

FUEL PROPERTIES

Due to the use of heavy water as both moderator and coolant, the CANDU reactor provides excellent neutron economy. On-line refuelling enables the CANDU reactors to run with a minimum of excess reactivity in the core. Together these two features of the CANDU reactor permit the use of a wide variety of fuel cycles, from the natural uranium fuel cycle used in currently operating plants, to enriched uranium and mixed-oxide cycles.

To meet the requirement imposed by the US Department of Energy to dispose of 50 tonnes of weapons grade plutonium over 25 years, the proposed MOX fuel must contain 1.5% fissile Pu, using the existing 37 element bundle design and with the maximum burnup within current limits established for natural uranium fuel.

A number of neutronic changes result from the use of MOX fuel. These include an overall increase in the reactivity of the lattice as well as a decrease in the reactivity depth of control devices. The neutron kinetics are affected as there is a decrease in the delayed neutron fraction and the prompt neutron lifetime. The negative temperature feedback is also decreased. These changes tend to make the MOX core more sensitive to reactivity perturbations unless compensating modifications are made to the fuel design or to the control and safety systems.

The flexibility of the CANDU design allows modifications to be made to the MOX fuel bundle design such that the MOX fuelled reactor operates within the current safety and licensing envelope. The major innovation of the MOX fuel design is the addition of a burnable poison to the inner elements of the bundle. This suppresses the excess reactivity of the MOX fuel, and reduces the coolant void reactivity. Dysprosium is used because its rate of depletion due to neutron capture matches those of the fissile elements. The amount of dysprosium added is sufficient to create a slightly negative void reactivity.

The 37-element bundle is composed of four rings of fuel elements containing the fuel pellets. There is a central element surrounded by rings of 6 elements, 12 elements and 18 elements in the outermost ring.

In the reference MOX design dysprosium is added to depleted uranium (0.2% U-235) in the inner two rings of the fuel bundle and plutonium is added to depleted uranium (0.2% U-235) in the outer two rings. The plutonium concentration in the outer ring of fuel is 1.2%, while that in the third ring of fuel is 2.0%. The reason for the increased plutonium concentration in the third ring is to compensate for the reduced neutron flux level due to the poisoning of the inner two rings of the fuel bundle, as well as for the flux depression in the inner rings of the bundle. This allows rings three and four, which produce virtually all of the power in the reference MOX design, to operate at similar power levels, thereby allowing a high-power level for the fuel, without exceeding the maximum allowable power rating for the individual fuel elements. The concentrations of the various fuel components are provided in Table 1.

As the two inner rings of fuel elements contain only the dysprosium poison and depleted uranium, the power ratings of these elements are very low. Thus the power produced by the bundle is essentially distributed over 30 elements rather than all 37. As a result, the overall bundle power limit must be decreased to maintain acceptable element power ratings. The use of a two-bundle-shift refuelling scheme in a Bruce A core without adjuster rods results in the flattening of the axial power distribution. As a result, the maximum channel and bundle powers in the MOX fuelled core are lower than those for the natural uranium core. The maximum channel and bundle powers for the MOX fuelled core are 7000 kW and 780 kW respectively, while those for the natural uranium core are 7200 kW and 950 kW respectively.

The maximum fuel element burnup of natural fuel elements in Bruce A is about 15,500 MWd/te, and this value is also used as the maximum burnup for the MOX reference fuel. The fuelling rate for the MOX fuel is 15.5 bundles per full power day, which is less than the fuelling rate of 18 bundles per day for natural fuel. A comparison of the k-infinity of natural fuel and that of the MOX reference fuel is presented in Figure 1. The MOX reference fuel initially has a higher reactivity than the natural fuel; however, in the absence of the buildup of plutonium, as is present in natural fuel, the reactivity decreases more quickly. The lattice code WIMS-AECL [1] was used to calculate lattice properties for the MOX fuel lattice.

An important consideration in the use of plutonium based fuel is that of the reactivity coefficients during power transients. This has been addressed by determining the reactivity change upon voiding as well as the reactivity change due to temperature changes in the fuel. The core-average void reactivity for natural fuel is 11 mk while the core-average void reactivity for the reference MOX fuel is -4.7 mk. The core-averaged fuel temperature coefficient of the MOX core: -3.0 μk/°C, is slightly less negative than that of the natural fuel core: -6.0 μk/°C.

Full core voiding of the MOX fuelled core is estimated to result in a net reactivity of -4.7 mk. This compares to +11 mk for the natural fuelled core. This inherent negative void reactivity of the MOX fuelled core results in an enhancement of the effectiveness of the safety systems which exist in the current CANDU core, and as a result, no hardware modifications to the existing control or safety systems would be required.

Table 2 lists the initial and discharge uranium and plutonium contents for the natural and MOX fuel bundles. As can be seen from the table, over half of the fissile plutonium is annihilated by the MOX fuel cycle. The discharged amounts of fissile material are similar for the natural fuel cycle and the MOX fuel cycle, even though the MOX fuel bundle enters the core with nearly twice the fissile material of the natural fuel bundle.

The MOX fuel core has been designed to operate within the current limits established for the natural uranium fuel core. A summary of the major characteristics of the reference MOX fuel core, as compared to those of the natural fuel core, is presented in Table 3.

FUEL MANAGEMENT

A 100-Full Power Day (FPD) refuelling simulation was carried out for a Bruce-A equilibrium MOX core using the 37-element fuel design. This was accomplished using the 3-D neutron diffusion code RFSP [2]. The simulation began with an equilibrium MOX core with a randomly chosen fuel distribution. The refuelling rate was 15.5 bundles per full power day. During this process the burnup and power of the bundles in the core, as well as those bundles discharged, was recorded for each simulation period of one full power day.

The distribution of the peak element burnup for discharged bundles in this simulation is displayed in Figure 2. The double peak of this distribution is due to the use of two burnup zones in the Bruce A core in order to flatten channel power distribution. This distribution compares favourably with data from discharged bundles from Bruce A using natural fuel. The median burnup is greater for the MOX fuel, due both to the higher average bundle burnup and the distribution of power over only 30 of the 37 fuel elements. However, the peak element burnups calculated in the simulation are less than those measured in Bruce A, because of the more uniform discharge fuel burnup distribution due to the used of the 2-bundle shift fuelling scheme.

Figure 3 illustrates the relationship between peak element rating and element burnup at an arbitrary time in the simulation. Each of the points in the figure gives the maximum element rating and corresponding element burnup for a single bundle in the core. As can be seen from the figure, the peak in this data is significantly less that the envelope for natural uranium fuel.

The maximum power boost, i.e. the change in power for MOX fuel elements after refuelling, was recorded for each simulation step for the 100 full-power-day simulation. For low-burnup bundles this is the result of power increase as the bundle is pushed towards the center of the core. For higher burnup bundles, the power increase is the result of the refuelling in adjacent channels. Figure 4 presents the envelope of power boost for varying burnup. The power boost envelope for MOX fuel is less than that for natural fuel (not shown) and is below the stress corrosion cracking threshold.

The results of the fuel management study therefore indicate that the channel powers, bundle powers, element ratings and refuelling ripples are all below the corresponding limits for the natural uranium core. The higher burnup rate of the MOX fuel allows the use of a two-bundle-shift fuelling scheme without exceeding the capacity of the existing fuelling machines. No additional risk of fuel failure due to the use of MOX fuel is expected.

NEUTRON KINETICS

In evaluating the safety of CANDU reactors it is necessary to evaluate the consequences of a Loss of Coolant Accident (LOCA). In natural uranium fuelled CANDU reactors, the reactivity of the core increases upon the loss of coolant. Rapid shutdown is required to limit the energy deposited in the fuel upon a LOCA.

For a given reactivity perturbation, the smaller delayed neutron fraction, and prompt neutron lifetime of the reference MOX fuel will result in more rapid response than for a natural uranium core. However, the inherent negative void reactivity of the MOX fuel core significantly improves the effectiveness of the safety systems during a LOCA, which is one of the most severe hypothetical accidents for CANDU reactors.

A point kinetic simulation was performed to compare the LOCA power transient in the natural fuel core with that in the MOX fuel core. The coolant density transient which was documented in the 1994 Bruce A Safety Report [3] is used for these simulations.

Fast neutronic signals initiate the reactor trip in the natural fuel core, because of the rapid power increase upon LOCA. However, in the reference MOX core, a LOCA will automatically result in a reduction of reactor power because of the inherent negative void reactivity. Therefore the relatively slow process system trip signals may be used to actuate the safety system. Available process trips, for a LOCA accident in a MOX fuelled core, include: Heat Transport System Low Pressure, Heat Transport System Low Flow, and Pressurizer Low Level. None of these process trips is anticipated to take longer than three seconds to initiate shutdown.

Simulated transients for a LOCA are plotted in Figure 5. As can be seen, the energy deposition in the MOX core is much less than that in the natural core for the same LOCA. The power deposited in the core for a natural-fuel core with shutdown on neutronic trips (0.13 seconds) is 4.35 full power seconds, while that for a MOX fuelled core with a three second trip delay is 2.35 full power seconds. The sensitivity of the power deposited in the fuel due to the process trip is investigated by varying the initiation of shutdown in the MOX fuelled core. The difference in the energy deposition as a result of a LOCA is 0.268 full power seconds or less than 15%, for a trip delay from 2 to 3 seconds. Hence, the LOCA transient in the MOX core is insensitive to the trip delay time.

Thus the absence of a LOCA power pulse greatly improves the effectiveness of the safety system. Evaluation of trip coverage concluded that the fuel can be adequately protected by process trips alone.

CONCLUSIONS

This study shows that two Bruce A reactors can be used to dispose of 50 tonnes of weapons-grade plutonium in 25 years, without requiring major modifications to the existing reactor design. The inherent safety feature that is engineered in the MOX fuel design enables the MOX-fuelled CANDU reactor to operate within the operational and safety limits of current natural uranium CANDU reactors.

REFERENCES

1 J.V. Donnelly, "WIMS-CRNL - A Users Manual for the Chalk River Version of WIMS", AECL-8955, January 1986.

2 D.A. Jenkins and B. Rouben, "Reactor Fuelling Simulation Program - RFSP: User's Manual for Microcomputer Version", Internal AECL Report, July 1993.

3 Ontario Hydro, "Bruce Generating Station A Safety Report", 1992.

Table 1: Comparison of Natural and MOX Fuel Compositions

	Natural	37 Element MOX
Matrix	Natural Uranium 0.71% U235 99.29% U238	Depleted Uranium 0.2% U235 99.8% U238
Ring 1 (1 pin)	-	5% Dysprosium
Ring 2 (6 pins)	-	5% Dysprosium
Ring 3 (12 pins)	-	2% Pu
Ring 4 (18 pins)	-	1.2% Pu

Table 2: Comparison of Initial and Exit Fissile Materials

	Natural Fuel (g/bundle)		37 Element MOX Fuel (g/bundle)	
	Initial	Discharge	Initial	Discharge
U-235	133.0	38.7	36.8	17.4
Pu-239	-	46.8	218.0	78.8
Pu-241	-	4.3	0.3	13.2
Total Fissile	133.0	89.8	255.1	109.4

Table 3: Comparison of Natural and MOX Core Characteristics

	Natural	37 Element MOX
Average Burnup (MWd/te)	8,300	9,700
Maximum Burnup (MWd/te)	15,000	15,500
Bundles/FPD	18	15.5
Fuel Management Scheme (bundles per channel refuelled)	2,4,8	2
Maximum Channel Power (MW)	7.200	7.000
Maximum Bundle Power (kW)	950	780
Average Bundle Fissile Content (wt%)	0.71	1.44

Table 4: Kinetic Data

	Natural	37 Element MOX
Full Core Void Reactivity (mk)	+11	-4.7
Fuel Temperature Coefficient (micro-k/degree C)	-6.0	-3.0
Total Delayed Neutron Fraction	0.00582	0.00383
Prompt Neutron Lifetime (Seconds)	0.0009	0.0005

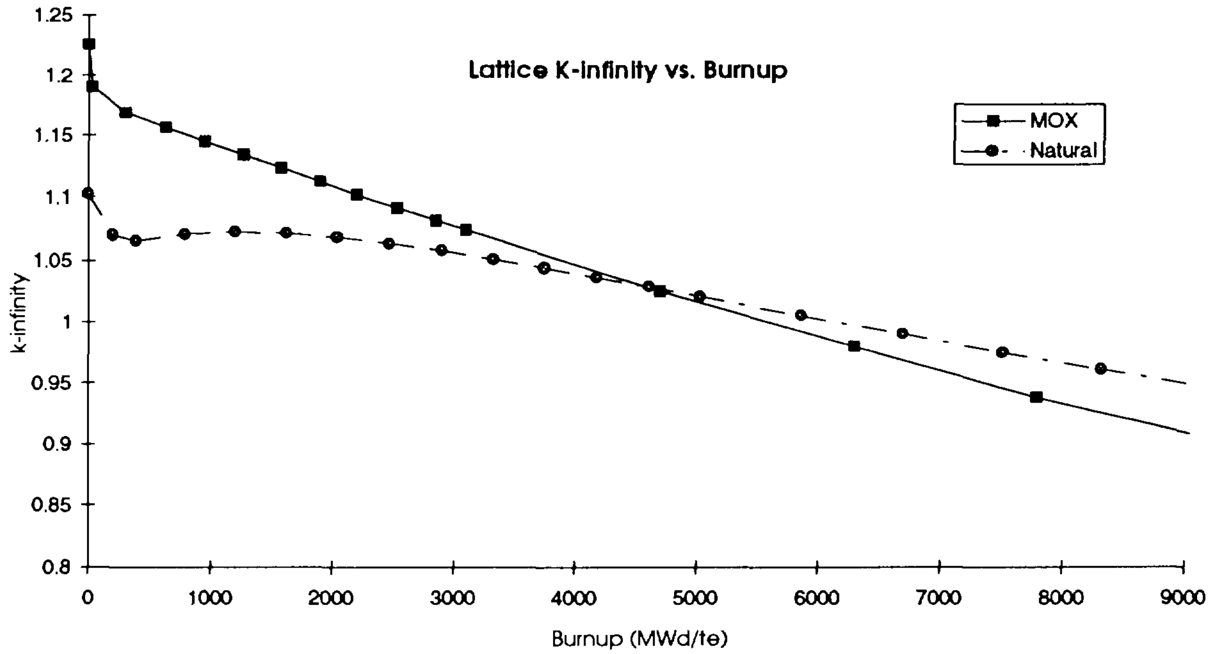

Figure 1: Lattice K-infinity vs. Burnup for the Reference Mox Fuel and Natural CANDU Fuel

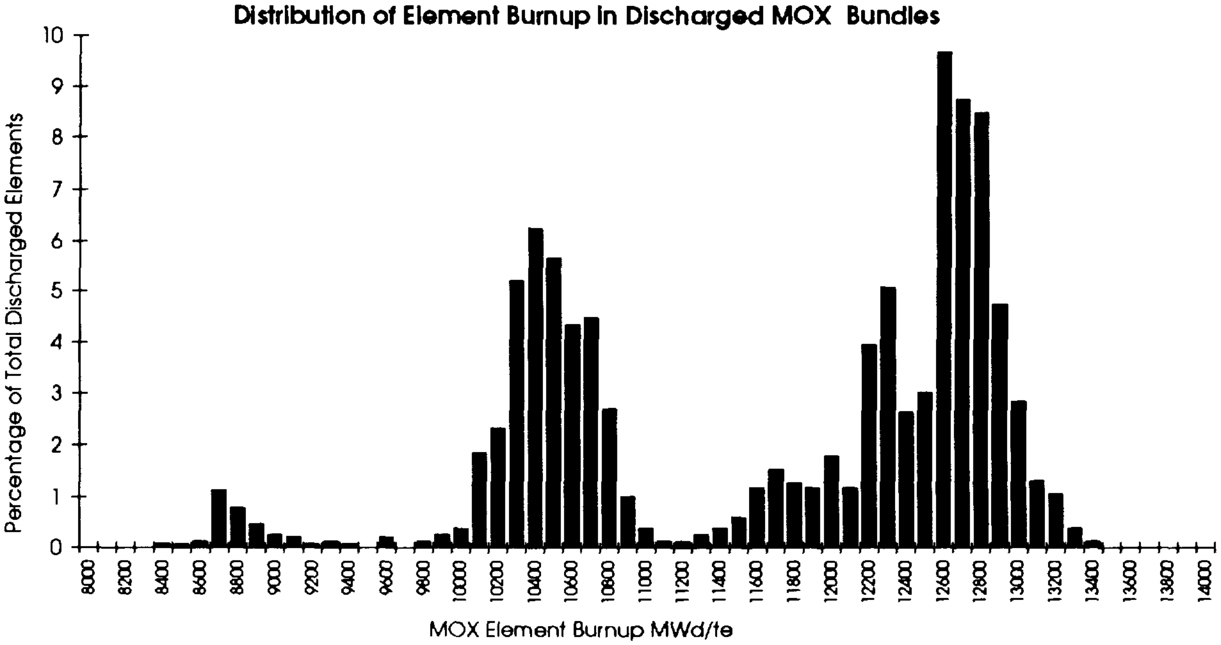

Figure 2: Distribution of Burnup in Discharged Reference MOX Bundles

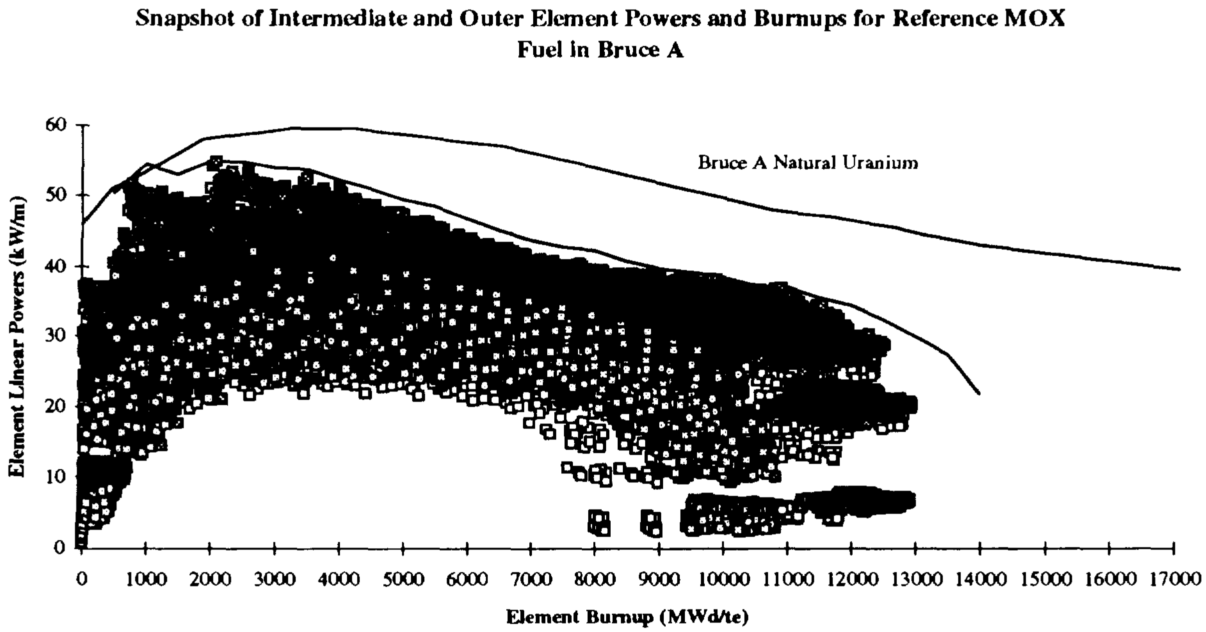

Figure 3: Maximum Element Rating vs. Burnup for Reference MOX Fuelled Core

SCC Power Increase Assessment for Reference MOX Fuel

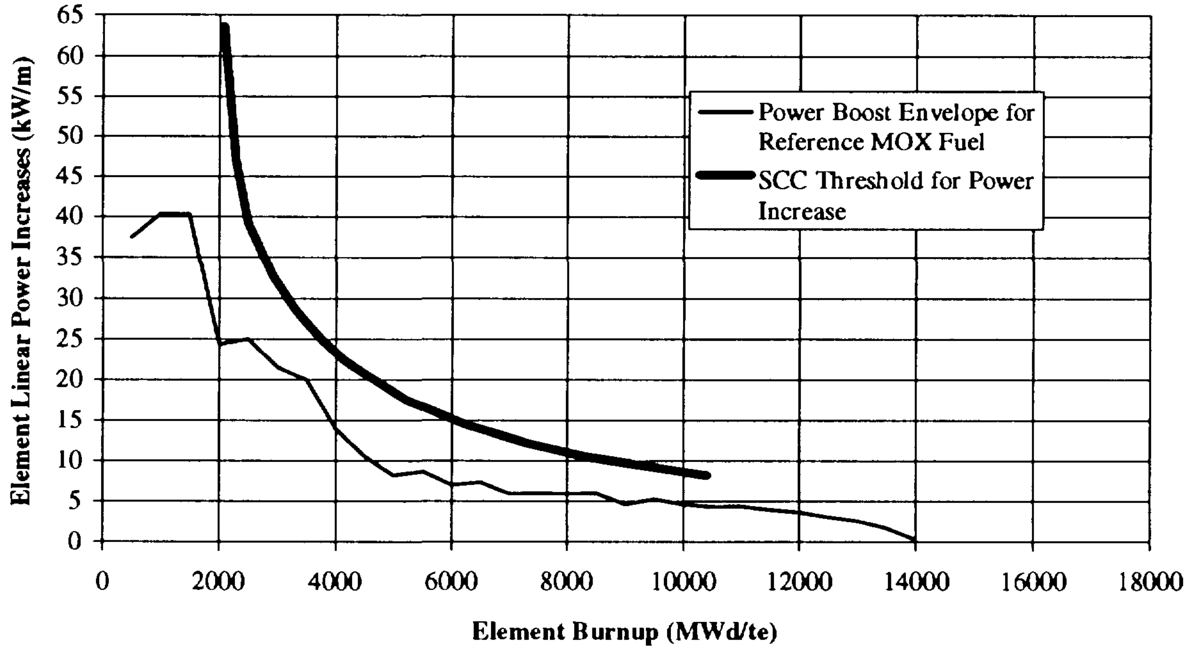

Figure 4: Maximum Element Rating vs. Burnup

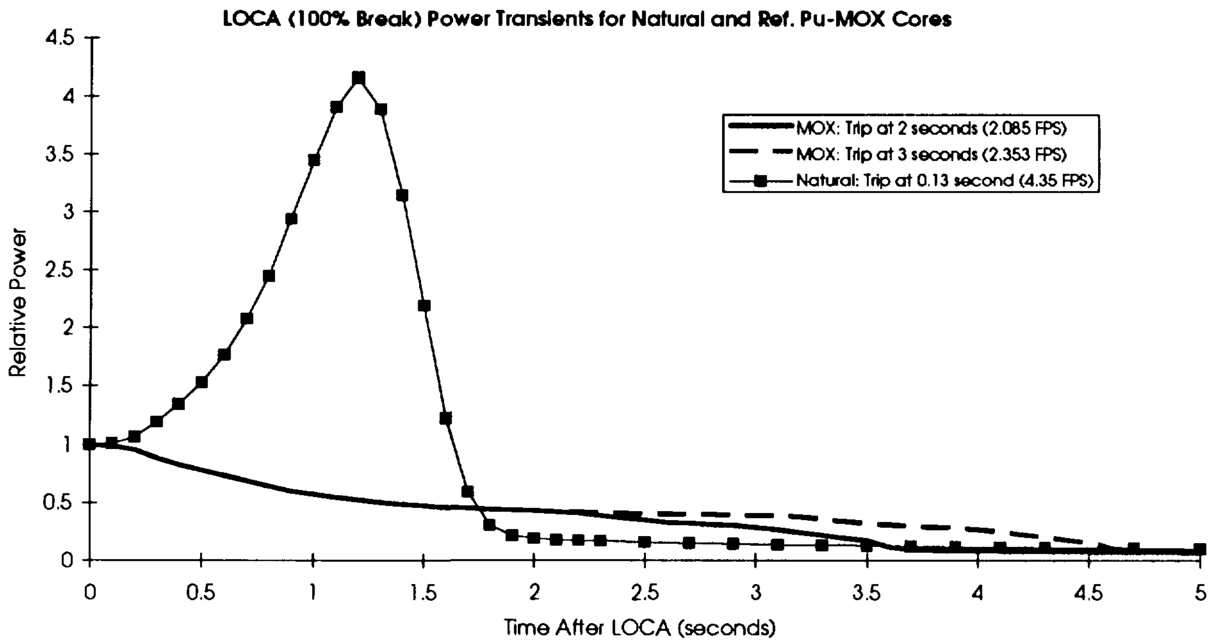

Figure 5: Comparison of LOCA Power Transients for Reference MOX Fuel and Natural CANDU Fuel